Tennessee Valley Authority, Post Office Box 2000, Soddy Daisy, Tennessee 37384-2000

April 25, 2013

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Unit 2
Facility Operating License No. DPR-79
NRC Docket No. 50-328

Subject: **Licensee Event Report 50-328/2013-001, "Manual Reactor Trip Due to Loss of Main Condenser Hotwell Level"**

The enclosed Licensee Event Report provides details concerning a manual reactor trip and automatic engineered safety feature actuation of the auxiliary feedwater system following a loss of main condenser vacuum indication and imminent loss of hotwell level. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in a manual or automatic actuation of the Reactor Protection System and the Auxiliary Feedwater System.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact Mr. Mike McBrearty, Sequoyah Site Licensing Manager, at (423) 843-7170.

Respectfully,

for

John T. Carlin
Site Vice President
Sequoyah Nuclear Plant

Enclosure: Licensee Event Report 50-328/2013-001
cc: NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resources@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant Unit 2	05000328	1 OF 9

4. TITLE:
Manual Reactor Trip Due to Loss of Hotwell Level

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									FACILITY NAME	DOCKET NUMBER
02	24	2013	2013 -	001 -	00	04	25	2013		

9. OPERATING MODE: 1

10. POWER LEVEL: 19%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Zachary Kitts	(423) 843-7018

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
A	SG	TBG	R369	Y	X	AB	LI	I204	N

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On February 24, 2013, at 1205 Eastern Standard Time, Operations personnel manually tripped the Sequoyah Unit 2 reactor from approximately 19 percent rated thermal power on imminent loss of condenser hotwell level. Specifically, Operations personnel entered an abnormal operating procedure based on indications of rising condenser pressure. When Operations personnel determined the hotwell level in “B” Condenser could not be maintained, a manual reactor trip was initiated. Prior to the manual reactor trip, the Unit 2 reactor had been at approximately 24 percent power with the main turbine offline for maintenance activities. All control rods fully inserted as required. The auxiliary feedwater system automatically initiated and provided feed water to the steam generators. No complications were experienced during or after the reactor trip. The loss of condenser vacuum indication was caused by the pressure indicator’s drain line breaking under cyclic fatigue. It is suspected that initial damage to the drain line was caused during the previous refueling outage. The drain line was replaced prior to restart of the unit. Corrective actions will be performed to provide shielding of the condenser pressure transmitters’ drain line, and the surrounding area will be marked as a unit trip hazard. Employees are to receive a briefing regarding the requirement for providing adequate spacing between stored items and sensitive equipment.

NARRATIVE

I. Plant Operating Conditions Before the Event

At the time of the event, Sequoyah Nuclear Plant (SQN) Unit 2 reactor was at approximately 24 percent rated thermal power (RTP) and the turbine was offline.

II. Description of Events

A. Event:

On February 24, 2013, at 1205 Eastern Standard Time (EST), Operations personnel manually tripped the reactor from approximately 19 percent rated thermal power (RTP) on imminent loss of condenser hotwell level. Specifically, Operations personnel entered an abnormal operating procedure based on indications of rising condenser pressure. When Operations personnel determined the hotwell level in “B” Condenser [EIIS Code SG] could not be maintained, a manual reactor trip was initiated.

The loss of condenser vacuum indication was caused by a failed condenser pressure instrument normally-closed drain line. The failure of the drain line resulted in the pressure instrument sensing a combination of condenser pressure and atmospheric pressure causing the pressure transmitter to indicate high condenser pressure. With high condenser pressure indicated, the steam dump control system [EIIS Code SB] logic closed the steam dumps resulting in the atmospheric relief valves (ARVs) [EIIS Code RV] opening to control steam pressure. This resulted in lowering hotwell levels and imminent loss of hotwell pumps and main feedwater pumps [EIIS Code SJ].

Just prior to the time of the event, SQN Unit 2 was operating at approximately 24 percent rated thermal power (RTP) using the condenser as the heat sink via steam dump control. Maintenance was being conducted on the “B” Low Pressure Turbine Rotor [EIIS Code TA]. The maintenance activity involved balancing the rotor in which a balance weight was installed through a designed port location of the “B” Condenser pressure boundary.

B. Status of structures, components, or systems that were inoperable at the start of the event and contributed to the event:

There were no inoperable structures, components or systems that contributed to this event.

NARRATIVE

C. Dates and approximate times of occurrences:

Dates and Times	Description
February 22, 2013 at 2300 EST	SQN Unit 2 is reduced from 100 percent RTP to conduct maintenance activities including balancing of the "B" Low Pressure Turbine Rotor.
February 23 at 0450	Operations placed steam dumps into service.
February 23 at 0546	The turbine is tripped and the steam dumps are controlling reactor power at approximately 24 percent RTP.
February 24 at 1159	Steam dumps closed after "B" Condenser vacuum indicated approximately 3.8 pounds per square inch atmospheric.
February 24 at 1205	Operations personnel tripped the reactor from approximately 19 percent RTP and took action to maintain the unit in Mode 3.
February 24 2013 following trip	"B" Condenser Pressure Instrument Loop, SQN-2-P-002-0007B, drain valve line, located on turbine building elevation 706 feet, was found severed.

D. Manufacturer and model number of each component that failed during the event:

1. The failed component was a 0.25 inch 316 stainless steel instrument sensing line drain line tubing [EIIS Code TBG]. This sensing line drain line tubing was connected to Rosemount, Model 1151 Alphaline Pressure Transmitter SQN-2-P-002-0007B [EIIS Code PT].

2. Number 1 Steam Generator (SG) [EIIS Code AB] Narrow Range Level Instrumentation Loop, SQN-2-L-003-0039 [EIIS Code LT]

E. Other systems or secondary functions affected:

No other systems or secondary functions were affected by this event.

NARRATIVE

F. Method of discovery of each component or system failure or procedural error:

<u>"B" Condenser Pressure Transmitter SQN-2-P-002-0007B instrument sensing line drain line tubing</u>

Discovery of the degrading condenser vacuum was self-revealing when the 2-PS-2-7B Condenser Vacuum Low alarm was received in the Main Control Room. The broken drain line for "B" Condenser Pressure Instrument was discovered in the field at the instrument location by Operations personnel.

<u>Number 1 Steam Generator Narrow Range Level Instrumentation Loop SQN-2-L-003-0039</u>

During the review of post-trip data, the dynamic response of the Number 1 SG Narrow Range Level Instrument Loop SQN-2-L-003-0039 was noted to have not responded properly during the loss of condenser steam dumps and the manual reactor trip. The instrument loop displayed an inconsistent level when compared to the other two redundant SG narrow range level indicators for the Number 1 SG for some period after the transients. This failure was not a contributor to the loss of condenser vacuum indication and subsequent manual reactor trip.

G. The failure mode, mechanism, and effect of each failed component, if known:

<u>"B" Condenser Pressure Transmitter SQN-2-P-002-0007B instrument sensing line drain line tubing</u>

The failed "B" Condenser Pressure Transmitter sensing line drain line was 0.25 inch 316 type stainless steel tubing. The drain line tubing failed during normal plant operation. The failure mechanism was cyclic fatigue resulting in fracture of the tubing. Fracture propagation had two initiation sites. The cause of the crack initiation sites are not conclusive, although it is believed to be the result of an external force that occurred during initial installation, maintenance, or other maintenance activity in close proximity. The failure of the drain line resulted in the pressure instrument sensing a combination of condenser pressure and atmospheric pressure causing the pressure transmitter to indicate high condenser pressure. With high condenser pressure indicated, the steam dump control system logic closed the steam dumps and ARVs opened to control steam pressure.

NARRATIVE

Number 1 Steam Generator Narrow Range Level Instrumentation Loop SQN-2-L-003-0039

This instrumentation loop displayed a level indication inconsistent with the other two safety grade narrow range level loops during the plant transients of losing condenser steam dump capability and the subsequent manual reactor trip.

The instrumentation loop was declared inoperable based on the review of the post-trip data and necessary actions of limiting condition for operation (LCO) 3.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” were entered. The failure mechanism was determined to be debris inside the wet reference leg of the sense line. The instrument loop responded correctly after the sensing line was blown down and the level reading was consistent with the other two narrow range loops. The loop was returned to operable status after testing.

The failure of the instrument loop did not adversely effect the start logic input it provides to the auxiliary feedwater (AFW) system [EIIS Code BA] because of the level instrumentation redundancy.

H. Operator actions:

Operations personnel responded to the decreasing condenser vacuum by entering Abnormal Operating Procedure (AOP) AOP-S.02, “Loss of Condenser Vacuum.” Operations personnel established a pre-determined condenser vacuum threshold at which the reactor would be manually tripped, if vacuum could not be restored. When Operations personnel determined condenser hotwell level could not be maintained, the reactor was manually tripped. Operations entered Emergency Procedure E-0, “Reactor Trip or Safety Injection” and subsequently transitioned to Emergency Subprocedure ES-0.1, “Reactor Trip Response”. When the plant was verified to be stable, Operations transitioned to 0-GO-6, “Power Reduction from 30% Reactor Power to Hot Standby.”

I. Automatically and manually initiated safety system responses:

The Reactor Protection System was manually initiated, with the safety systems responding to the reactor trip as designed. The AFW system started and maintained steam generator levels as expected. One of three level indication instrument loops for the Number 1 SG was determined to be inoperable after the reactor trip, as discussed previously.

NARRATIVE

III. Cause of the event

A. The cause of each component or system failure or personnel error, if known:

<u>"B" Condenser Pressure Transmitter SQN-2-P-002-0007B instrument sensing line drain line tubing</u>

The "B" Condenser Pressure Transmitter sensing line drain line failure mechanism was cyclic fatigue resulting in fracture of the tubing. Fracture propagation had two crack initiation sites. The pressure transmitter is coupled to the condenser wall, which experiences forced vibration during operation. The cause of the crack initiations are not conclusive, although it is believed to be the result of an external force that occurred during initial installation, maintenance, or other maintenance activity in close proximity. Although not conclusive, the external force may have been caused by something striking the sensing line drain line as maintenance was conducted. Material was reported being stored in close proximity during the previous refueling outage of Fall 2012. Plant procedures provide guidance in regards to spacing for working and storage of materiel near sensitive equipment.

<u>Number 1 Steam Generator Narrow Range Level Instrumentation Loop SQN-2-L-003-0039</u>

The cause of the level instrumentation loop failure was determined to be debris inside the wet reference leg of the sense line. The removed debris was determined to be magnetite, a ferrous iron oxide.

B. The cause(s) and circumstances for each human performance related root cause:

The root cause of the failed drain line may have had human performance related aspects. It could not be absolutely confirmed that employees, during the previous refueling outage, caused damage to the sensing line drain line.

IV. Analysis of the event:

Unit 2 was being maintained in Mode 1 at approximately 24 percent RTP. Prior to the steam dump closure, the Reactor Coolant System (RCS) [EIIS Code AB] temperature was at approximately 553 degrees Fahrenheit (F) and pressure was approximately 2224 pounds per square inch gauge (psig). The turbine generator was off-line for maintenance activities. The condenser was in operation as the heat

NARRATIVE

sink via the steam dump control, with steam pressure at approximately 950 psig. After failure of the "B" Condenser sensing line drain line and closure of the steam dumps, reactor power lowered to 19 percent RTP, RCS temperature and pressure increased to approximately 563 degrees F and 2265 psig, respectively. The pressurizer level had risen to 46 percent based on the RCS temperature change due to the steam transient. Both the motor driven and the turbine driven AFW pumps and the ARVs were available. At approximately 1205 EST the reactor was manually tripped. RCS temperature decreased to a minimum value of approximately 545 degrees F and remained within technical specifications (TSs) limits. The minimum RCS pressure following the reactor trip was approximately 2088 psig, which is well above the pressure that would have initiated a safety injection signal (1870 psig). The decrease in reactor coolant temperature and volume resulted in the rapid decrease in pressurizer level to approximately 28 percent and then stabilization near program value. The plant response was expected because of the low initial power level and low decay heat as the plant was near beginning of core life, less than two months from end of the Fall 2012 refueling outage. No TS safety limits were exceeded and the Updated Final Safety Analysis Report (UFSAR) analyses remained bounding.

V. Assessment of Safety Consequences

A. Availability of systems or components that could have performed the same function as the components and systems that failed during the event:

Safety-related systems that were needed to shut down the reactor, maintain safe shutdown conditions, remove residual heat or mitigate the consequences of an accident remained available throughout the event.

B. For events that occurred when the reactor was shut down, availability of systems or components needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident:

This event did not occur when the reactor was shut down. Safety-related systems that were needed to shut down the reactor, maintain safe shutdown conditions, remove residual heat or mitigate the consequences of an accident remained available throughout the event.

C. For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from discovery of the failure until the train was returned to service:

NARRATIVE

There was no failure that rendered a train of a safety system inoperable during this event.

VI. Corrective Actions

Corrective Actions are being managed by TVA's Corrective Action Program under PER 686710 and 691935.

A. Immediate Corrective Actions:

A walk down of condenser pressure transmitters and sense lines was conducted for both Unit 1 and Unit 2. In Unit 2, Pressure Transmitter SQN-2-PT-002-0010 for the "C" Condenser was found to have a bent instrument sensing line drain line. This deficiency was entered into the site's Corrective Action Program and subsequently corrected prior to plant startup. The remaining Unit 2 pressure transmitters were found in acceptable condition. The Unit 1 condenser pressure transmitters were found to be in acceptable condition.

B. Corrective Actions to prevent recurrence or to reduce probability of similar events occurring in the future:

1. Protect the condenser pressure transmitter drain lines by installing a protective guard around the drain line and marking the area surrounding the instrument line as a unit trip hazard.

2. A sample review is to be conducted of various instrument loops to determine if 1) the sensing line has tubing; 2) the failure of the tubing instrument line would cause a plant transient, manual reactor trip, or automatic reactor trip; and 3) developed inspection criteria for the acceptability of the instrument lines and conduct physical inspection of tubing instrument lines using the criteria.

3. Provide briefing to employees regarding the requirement for providing adequate spacing between stored items and sensitive equipment.

VII. Additional Information

A. Previous similar events at the same plant:

SQN Problem Evaluation Report SQ972542PER, dated November 5, 1997:
This evaluation report documents a condition where the sensing line to condenser Pressure Transmitter 2-PT-2-7 was broken and isolated steam dumps to the condenser due to sensed high condenser pressure. The cause of the event was

NARRATIVE

failure of the 0.25 inch sensing line tubing under cyclic fatigue. The location equipment panels were changed to reduce vibration. The change also increased the sensing line tube to 0.375 inch; however, the sensing line drain line size remained at 0.25 inch.

B. Additional Information:

None.

C. Safety System Functional Failure Consideration:

This event did not result in a safety system functional failure.

D. Scrams with Complications Consideration:

This event did not result in an unplanned scram with complications.

VIII. Commitments:

None.